NRC/Framatome Meeting on the MOX Fuel Design Report BAW-10238(NP)

June 18, 2002

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MOX Fuel Design Report Meeting - Agenda

> Background and orientation
> Purpose
> MOX design considerations
> Weapons grade plutonium
> Weapons derived plutonium
> Manufacturing processes
> Mark-BW/MOX1 design description
> Mark-BW/MOX1 design evaluation
> Experience base
> Lead assembly program
> Conclusions

MOX Fuel Design Report - Background

- Plutonium disposition program
- MOX fuel project
- Fuel qualification
- MOX fuel related submittals

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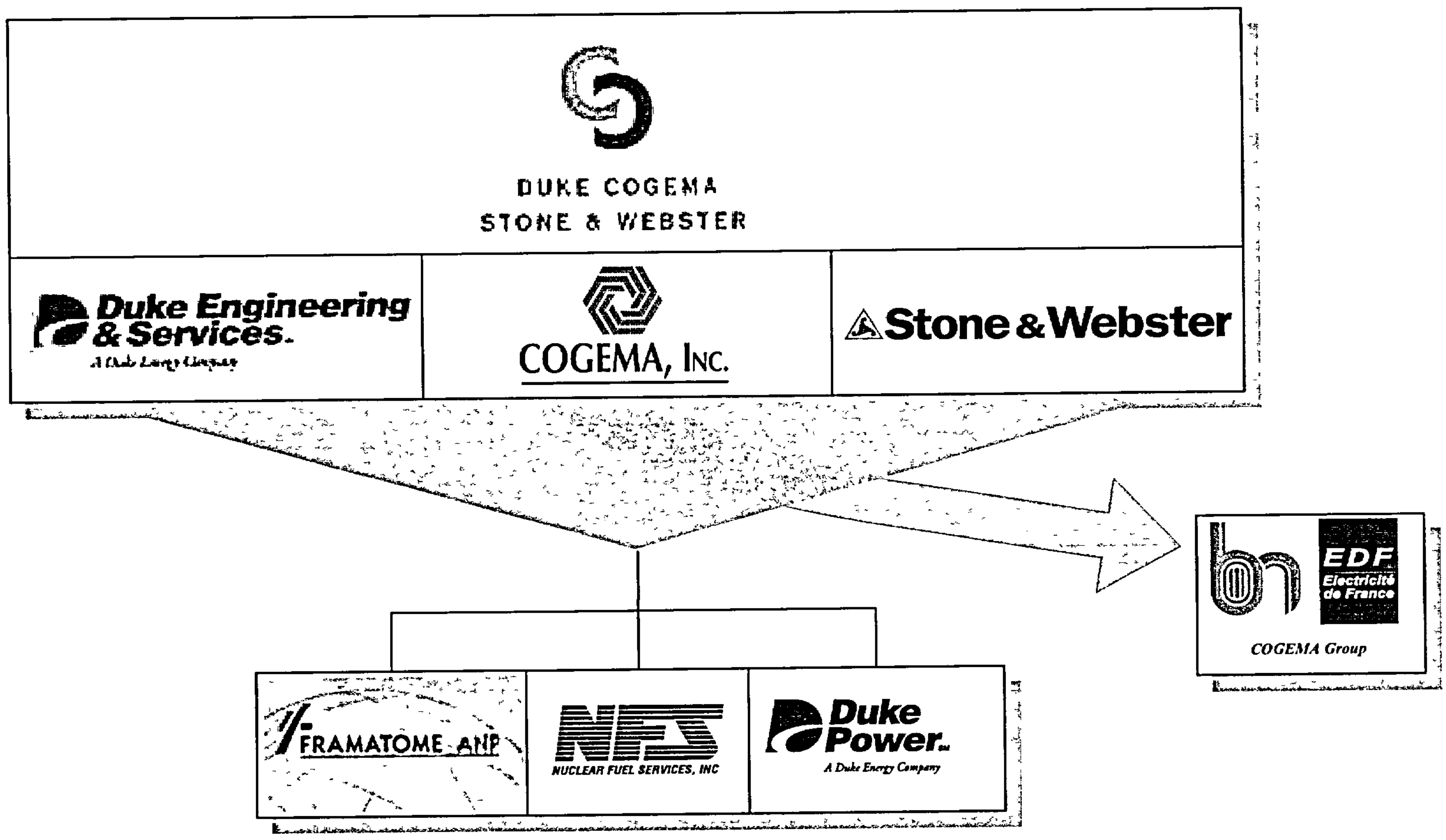
Plutonium Disposition Program

- Goal: To dispose of surplus weapons plutonium
  - January 2000 Department of Energy (DOE) Record of Decision
  - September 2000 U.S.-Russian Federation Plutonium Disposition Agreement
- Approach
  - Fabrication into mixed oxide (MOX) fuel and use in existing light water reactors; immobilization in vitrified high level waste
- Recent Changes
  - Termination of immobilization
  - Design changes to the MOX Fuel Fabrication Facility to accommodate a wider variation of feed material
  - One year delay in the provision of batch quantities of MOX fuel from the MOX Fuel Fabrication Facility - 2008

Plutonium Disposition Program

Duke Cogema Stone & Webster (DCS) Team



MOX Fuel Project

> MOX Fuel Fabrication

- 12/00: DCS submitted Environmental Report to NRC
- 2/01: DCS submitted Construction Authorization Request to NRC

> MOX Fuel Qualification

> MOX Fuel Irradiation

> Fresh MOX Fuel Transportation and Packaging

> Project Management

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MOX Fuel Qualification and Irradiation

> Fuel Qualification Approach

- Maximize use of European experience base
  - Research programs
  - Established manufacturing process
  - Reactor irradiation experience
- Utilize a proven fuel assembly design
- Confirm performance with lead assembly program

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MOX Fuel-Related Submittals

> July 2000: DCS Fuel Qualification Plan provided to NRC for information

> August 2000: Framatome COPERNIC Topical Report (MOX applications)

> April 2001: DCS MOX Fuel Qualification Plan revised and provided to NRC for information

> August 2001: Duke Power Nuclear Analysis Topical Report (MOX and LEU applications)

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MOX Fuel-Related Submittals (cont.)

> September 2001: Duke Power Thermal-Hydraulic Statistical Core Design Topical Report, Appendix E (advanced Mk-BW fuel assembly design, to be used for MOX fuel)

> April 2002: Framatome Advanced Mark-BW Fuel Assembly Design Topical Report

> April 2002: Framatome MOX Fuel Design Report – BAW-10238(NP)

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MOX Fuel Qualification and Irradiation Plans

> 2002?: Submit MOX Fuel Lead Assembly License Amendment Request (Duke Power)

> 2003?: Submit Updated Fuel Qualification Plan (DCS)

> 2003: Submit MOX Fuel Safety Analysis Topical Report (Duke Power)

> December 2003: Submit License Amendment Requests for Batch Utilization of MOX Fuel at McGuire and Catawba (Duke Power)

> 2004: Submit MOX Fuel LOCA Topical Report (Framatome)

> 2004?: Begin MOX fuel lead assembly irradiation

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MOX Fuel Design Report - Purpose

- Confirm safe and reliable operation of fuel used for Material Disposition Program – Mark-BW/MOX1
- Demonstrate that Mark-BW/MOX1 is acceptable for batch implementation up to a maximum fuel rod burnup of 50,000 MWd/MThm

MOX Fuel Design Report – Design Considerations

> Performance characteristics

- Material properties – Included in COPERNIC (BAW-10231P)
  - Thermal conductivity
  - Thermal expansion
  - Thermal creep
  - Fission gas release
  - In-reactor densification and swelling
  - Helium gas accumulation and release
  - Radial power profile
  - Melting point
- MOX specific models for thermal conductivity, fission gas release, radial power profile, and melting point

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MOX Fuel Design Report – Design Considerations

> Isotopics

- At BOL, fissionable component is ^{239}Pu rather than ^{235}U
- At EOL, uranium based fuel produces approximately 40% of its power from plutonium

Isotope	BOL		EOL	
	Uranium Fuel	MOX Fuel	Uranium Fuel	MOX Fuel
^{234}U	0.04	0.00	0.02	0.00
^{235}U	4.60	0.24	0.82	0.09
^{236}U	--	--	0.62	0.03
^{238}U	95.36	95.39	91.51	92.28
^{238}Pu	--	0.00	0.04	0.02
^{239}Pu	--	4.08	0.65	1.39
^{240}Pu	--	0.29	0.28	0.85
^{241}Pu	--	0.00	0.19	0.50
^{242}Pu	--	0.00	0.09	0.16
^{241}Am	--	0.00	0.01	0.02

MOX Fuel Design Report – Design Considerations

> Pellet homogeneity and microstructure

- LEU fuel homogeneity ensured due to enrichment process
- MOX manufacturing involves blending and milling of UO_2 and PuO_2 powders
- MOX and UO_2 are comparable on macro-scale
- Differences on micro-scale – MOX pellets contain plutonium finely dispersed in UO_2 matrix and micron size islands of plutonium rich particles
  - Particles are not pure plutonium
  - Particles are plutonium rich Master Mix particles – plutonium content derived from ratio of UO_2 to PuO_2 in the Master Mix

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MOX Fuel Design Report – Design Considerations

> Pellet homogeneity and microstructure (continued)

- Maximum size and plutonium content
  - Determined by manufacturing process
  - Milling and sieving operations, followed by sintering process that induces diffusion of PuO_2 bearing particles into the UO_2 lattice
  - Control of process verified through metallographic examinations
- Pellet specification limits average and maximum sizes of plutonium rich agglomerates
  - Mean particle size less than 50 microns
  - Maximum particle size less than 400 microns
- Identical specifications for European RG MOX and DCS/FRA WG MOX

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MOX Fuel Design Report – Weapons Grade Plutonium

> Plutonium defined as Weapons Grade or Reactor Grade based on isotopics

Plutonium Isotope	Weapons Grade	Reactor Grade
^{238}Pu	0.0	1.0
^{239}Pu	93.6	59.0
^{240}Pu	5.9	24.0
^{241}Pu	0.4	10.0
^{242}Pu	0.1	5.0
^{241}Am	0.0	1.0

MOX Fuel Design Report – Weapons Grade Plutonium (continued)

> Higher fissile content of WG material allows lower plutonium concentrations
  - MOX fuel from RG material may be 8% to 9%
  - MOX fuel from WG material for MD program will be less than 6%

> WG material has lower neutron dose

> WG material has lower gamma dose and lower heating

> Neutronic modeling of all uranium cores requires capability to model plutonium

> RG MOX fuel more challenge to modeling than WG MOX

> WG MOX fuel characteristics bounded by LEU fuel and RG MOX

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MOX Fuel Design Report – Weapons Derived Plutonium

> WG MOX isotopics
> May contain impurities - most significant impurity is gallium
> Gallium concerns
  - Processing equipment – furnaces
  - In-reactor performance – effect on cladding
> Gallium will be effectively eliminated from the plutonium used in the MD program
  - Aqueous polishing process to be used at the MOX Fuel Fabrication Facility

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MOX Fuel Design Report – Weapons Derived Plutonium (continued)

> Gallium in weapons derived plutonium

- Effectiveness of aqueous polishing
  - Decontamination factor (DF) greater than 10^5
  - Maximum gallium concentration in weapons derived Pu – 1.2%
  - Maximum gallium concentration in feed powder - .12 ppm
  - Resulting gallium concentration in MOX pellet from weapons derived Pu - .006 ppm

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MOX Fuel Design Report – Weapons Derived Plutonium (continued)

■ Gallium Concentration in Archive Fuel Pellets

Unit	Fuel Type	Pellet Vendor	Nominal Enrichment (^{235}U)	Date of Manufacture	Pellet Gallium Content (Avg. 5 samples) (ppb)
Catawba Unit 1	Mark-BW (17x17)	General Electric	3.55%	October 1990	9.8
McGuire Unit 2	Mark-BW (17x17)	Siemens	3.65%	December 1992	11.5
TMI	Mark-B (15x15)	Siemens	4.75%	June 1993	9.0
Davis Besse	Mark-B (15x15)	Siemens	3.79%	May 1994	10.8

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MOX Fuel Design Report – Weapons Derived Plutonium (continued)

- Gallium Concentration in Archive Fuel Components

Component	Number of Samples	Average Gallium Content
Plenum Spring	9	38,200 ppb
Zircaloy-4 Cladding	6	275 ppb

MOX Fuel Design Report – Weapons Derived Plutonium (continued)

> Fuel performance with gallium

- ATR test program
  - Began irradiation in January 1998
- Two types of MOX fuel
  - Untreated weapons derived plutonium with 3.0 ppm gallium in the MOX pellets
  - Thermally treated MOX with 1.3 ppm gallium
- Test rods operating at 5 – 10 kw/ft
- Test rods extracted at 8,000 MWd/MThm; 21,000 MWd/MThm; 30,000 MWd/MThm
- Projected burnup of 50,000 MWd/MThm
- Hot cell examinations have been performed; at 30,000 MWd/MThm -
  - SEM/microprobe examinations of fuel and cladding revealed no abnormal behavior
  - No indication of gallium migration to the cladding;analyses of unirradiated cladding indicates no transfer of gallium from the fuel to the cladding

MOX Fuel Design Report – Weapons Derived Plutonium (continued)

> Gallium – summary

- MD program will use aqueous polishing to remove essentially all of the gallium
- Plutonium powder specification will limit gallium to trace levels (tenths of ppm)
- Resulting MOX pellets will be in the low ppb range
- Uranium fuels have operated successfully with comparable and higher levels of gallium
- Test rods are operating successfully at representative burnups and heat rates with gallium levels much higher than the MD program will use

MOX Fuel Design Report – Weapons Grade Plutonium

> Plutonium concentrations lower with WG plutonium

> Pellet Microstructure

- Use of WG plutonium in MOX has potential to affect:
  - Thermal conductivity
  - Fission gas release
  - Fuel pellet swelling
  - Radial power distribution

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MOX Fuel Design Report – Weapons Grade Plutonium

> Pellet microstructure for WG MOX is equivalent to RG MOX

- UO_2 matrix establishes overall pellet microstructure – RG MOX and WG MOX use same feed UO_2
- Grain size, particle size, particle distribution same for RG MOX and WG MOX
- Plutonium rich agglomerates equivalent in fissile content with RG fuel
  - Master Mix for RG plutonium uses UO_2/PuO_2 ratio of 70/30
  - Master Mix for WG plutonium will use UO_2/PuO_2 ratio of 80/20
- Distribution of fissile material is same for RG MOX and WG MOX
- Fission density and fission product inventory the same for RG MOX and WG MOX

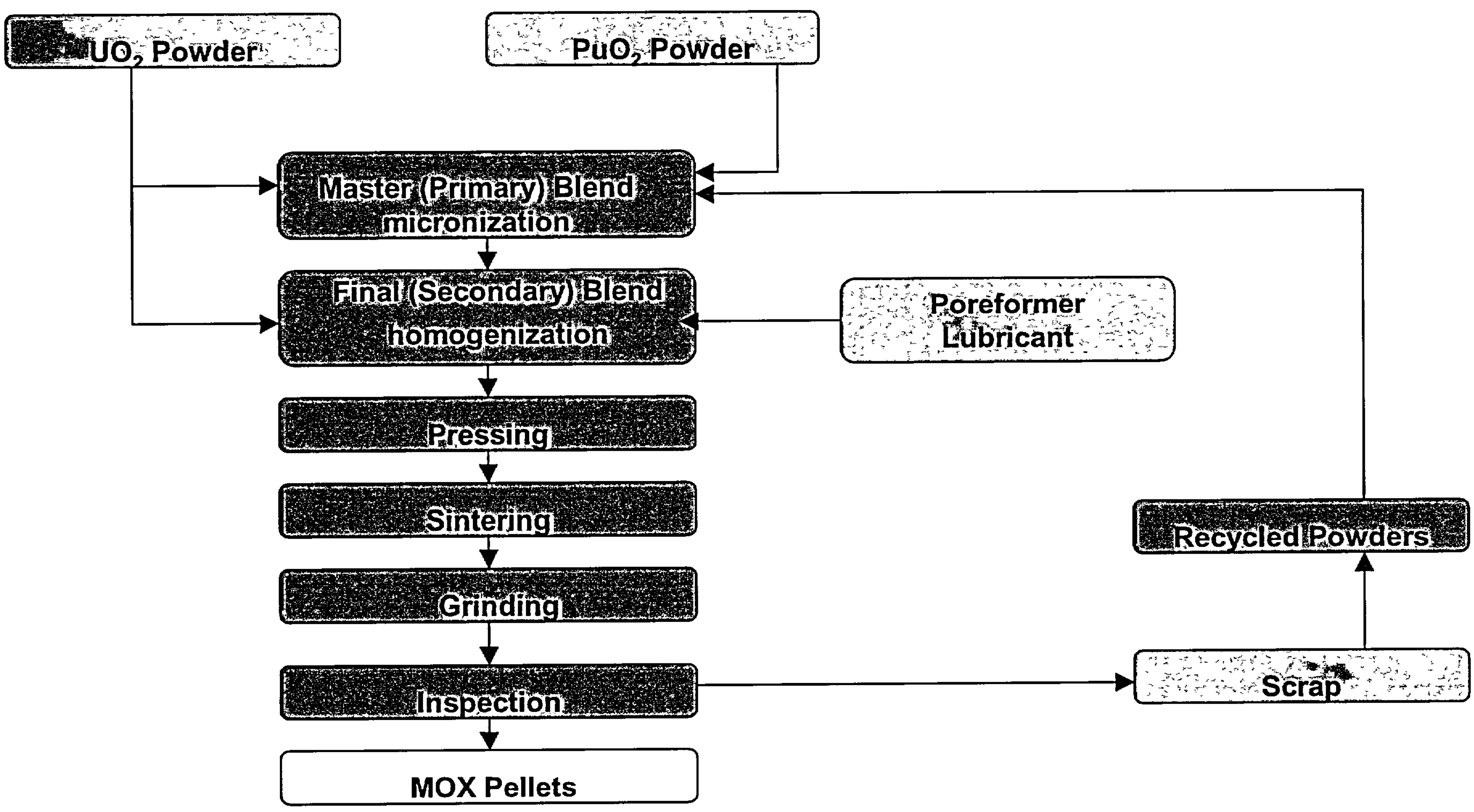
> Properties for WG MOX same as, or bounded by, RG MOX

MOX Fuel Design Report – Manufacturing

FABRICATION OF MOX PELLETS

MIMAS PROCESS

"MIcronized MASter blend"



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MOX Fuel Design Report – Mark-BW/MOX1 Design Description

> MOX program will utilize the proven Mark-BW as the structural foundation for the MOX fuel irradiation

> The base Mark-BW, with mid-span mixing grids and M5 materials, is designated 'Advanced Mark-BW' and is detailed in BAW-10239P

> For MOX applications, the Advanced Mark-BW is termed 'Mark-BW/MOX1'

> Advanced Mark-BW and Mark-BW/MOX1 features:

- Seated fuel rods
- Floating intermediate spacer grids
- Removable top nozzle
- Trapper™ bottom nozzle
- M5™ alloy cladding and structural materials

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MOX Fuel Design Report – Mark-BW/MOX1 Design Description

> MOX Fuel Rod Design

- MOX pellets utilizing European specification for RG MOX (with the addition of requirements for gallium)
  - Stack length – 144 inch
  - 95% TD
  - 0.3225 inch diameter
- Cladding
  - 0.374 inch OD
  - 0.0225 inch wall thickness
- Rod – increased length for fission gas release

MOX Fuel Design Report – Mark-BW/MOX1 Design Description

> Mark-BW/MOX1 Design Comparison

Parameter	Advanced Mark-BW Value	Mark-BW/MOX1 Value
Pellets		
Fuel Pellet Material	Enriched UO_2	PuO_2 and Depleted UO_2
Fuel Pellet Diameter, in.	0.3225	0.3225
Fuel Pellet Theoretical Density, %T.D.	96	95
Fuel Pellet Volume Reduction due to Chamfer and Dish, %	1.24	1.11
Rods		
Fuel Rod Length, in.	152.16	152.40
Fuel Rod Cladding Material	M5™	M5™
Fuel Rod Inside Diameter, in.	0.329	0.329
Fuel Rod Outside Diameter, in.	0.374	0.374
Active Fuel Stack Height, in.	144	144
Maximum Fuel Rod Burnup, MWd/MThm	60,000	50,000

MOX Fuel Design Report – Mark-BW/MOX1 Design Description

>Mark-BW/MOX1 Design Comparison

Parameter	Advanced Mark-BW Value	Mark-BW/MOX1 Value
Assemblies		
Fuel Assembly Length, in.	159.8	159.8
Lattice Geometry	17x17	17x17
Fuel Rod Pitch, in.	0.496	0.496
Number of Fuel Rods per Assembly	264	264
Heavy Metal Loading per Assembly, kg	466.1	462.8
Number of Grids		
Bottom End	1	1
Vaneless Intermediate	1	1
Vaned Intermediate	5	5
Mid-Span Mixing	3	3
Top End	1	1

MOX Fuel Design Report – Mark-BW/MOX1 Design Evaluation

- >Meets all applicable criteria
- >Mechanical analyses satisfies requirements of Section 4.2 of SRP
- >All interfaces with resident fuel assemblies and internals are preserved
- >Analyses of BAW-10239P apply to Mark-BW/MOX1
  - Analyses affected by pellet characteristics have been re-evaluated
  - Fuel rod analyses follow previously approved methods but use COPERNIC (BAW-10231P) with MOX specific models

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MOX Fuel Design Report – Mark-BW/MOX1 Design Evaluation

> Evaluations

- Fuel System Damage
  - Stress
    - Fuel Assembly Stress – Fuel rod increases only 0.25 inch; stress analyses in BAW-12039P remain valid for Mark-BW/MOX1
    - Cladding Stress – Evaluated with NRC approved methodology for M5; margins comparable to LEU fuel were confirmed
  - Cladding Strain – Transient strain calculated with COPERNIC; margins comparable to LEU fuel
  - Cladding Fatigue – NRC approved methodology used; margins comparable to LEU fuel
  - Fretting – Evaluation methods in BAW-10239P are applicable
  - Oxidation, Hydriding – COPERNIC used to evaluate corrosion for M5 cladding with NRC approved methods; margins comparable to LEU fuel
  - Fuel Rod Bow – Evaluation methods in BAW-10239P are applicable
  - Axial Growth – Effects of increased fast neutron fluence included; positive shoulder gap maintained throughout life
  - Fuel Rod EOL Pressure – COPERNIC used with NRC approved methodology; fuel rod internal pressure criterion is met
  - Assembly Liftoff – Evaluation methods in BAW-10239P are applicable

MOX Fuel Design Report – Mark-BW/MOX1 Design Evaluation

> Evaluations (continued)

- Fuel Rod Failure
  - Internal Hydriding – Pellet and fuel rod specifications control total moisture content
  - Creep Collapse – NRC approved methodology used; margins comparable to LEU fuel
  - Overheating of Cladding – DNB performance not affected by changes to accommodate MOX pellets; Duke will perform analyses with NRC approved methods
  - Overheating of Fuel Pellets – Minor thermal conductivity decrease relative to LWU fuel; COPERNIC used to establish limits comparable to LEU fuel
  - Pellet/Cladding Interaction – Cladding strain and fuel melt criteria ensure PCI is addressed
  - Cladding Rupture – LOCA analyses addressed in separate report

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MOX Fuel Design Report – Mark-BW/MOX1 Design Evaluation

> Evaluations (continued)

- Fuel Coolability
  - Cladding Embrittlement – LOCA analyses provided in separate report
  - Violent Expulsion of Fuel – Duke will submit safety analysis evaluations, including reactivity insertion accidents
  - Fuel Rod Ballooning – LOCA analyses provided in separate report
  - Fuel Assembly Structural Damage from External Forces – Analyses documented in BAW-10239P are bounding

MOX Fuel Design Report – Experience Base

> Framatome ANP U.S. Experience

- FRA-ANP Mark-BW fuel has operated in all four mission reactors
- FRA-ANP has supplied over 2,500 Mark-BW fuel assemblies to U.S. 17x17 reactors
- Mark-BW has experienced a failure rate of less than one per 100,000 rods, from all manufacturing related causes, since first supplied in 1987

MOX Fuel Design Report – Experience Base

No.	Country	Reactor	MELOX	Cadarache	Dessel
1		Blayais 1	X		
2		Blayais 2	X	X	X
3		Dampierre 1	X	X	X
4		Dampierre 2	X	X	X
5		Dampierre 3	X		
6		Dampierre 4	X		
7		Tricastin 1	X		
8		Tricastin 2	X	X	
9		Tricastin 3	X	X	
10		Tricastin 4	X		
11	France	St. Laurent 1	X	X	X
12		St. Laurent 2	X	X	X
13		Gravelines 1	X		
14		Gravelines 2	X		
15		Gravelines 3	X	X	X
16		Gravelines 4	X	X	X
17		Chinon 1	X		
18		Chinon 2	X		
19		Chinon 3	X		
20		Chinon 4	X		
21	Belgium	Tihange 2			X
22		Doel 3			X
23		Unterweser		X	X
24		Grafenrheinfeld			X
25		Phillipsburg 2		X	X
26		Brokdorf			X
27		Gundremmingen B			X
28	Germany	Gundremmingen C			X
29		Grohnde		X	
30		Isar 2		X	
31		Obrigheim		X	
32		Neckarwestheim 2		X	
33		Beznau 1			X
34	Switzerland	Beznau 2			X
35		Gosgen			X

MOX Fuel Design Report – Experience Base

> European experience

- Fuel reliability – 435,000 operating MOX fuel rods
  - Six (6) total failures
  - Five (5) confirmed debris
  - One (1) suspected debris
- Use of MIMAS process, with polished Pu, ensures that European experience base is applicable to WG Pu used in the MD program
- Experimental data –
  - Used in model development and benchmarking
  - Included in COPERNIC submittal (BAW-10231P)
  - Operating data included in Duke Physics Topical

MOX Fuel Design Report – Experience Base

> European experience

Country	Reactors		Maximum Discharge Burnups (MWd/MThm) of Assemblies Having Completed		
	Number	Type	3 Cycles	4 Cycles	5 Cycles
Framatome ANP, SSA (France) Deliveries					
France	20	17 x 17	40,500	46,000	55,000 (61,000 - Rod)
Belgium	2	17 x 17	44,000	46,500	
Germany	2	16 x 16 18 x 18	43,000	52,000	
Framatome ANP, GmbH (formerly Siemens) Deliveries					
Germany	9	14 x 14 to 18 x 18		49,000	
Switzerland	3	14 x 14 and 15 x 15		54,000 (65,000 - Rod)	

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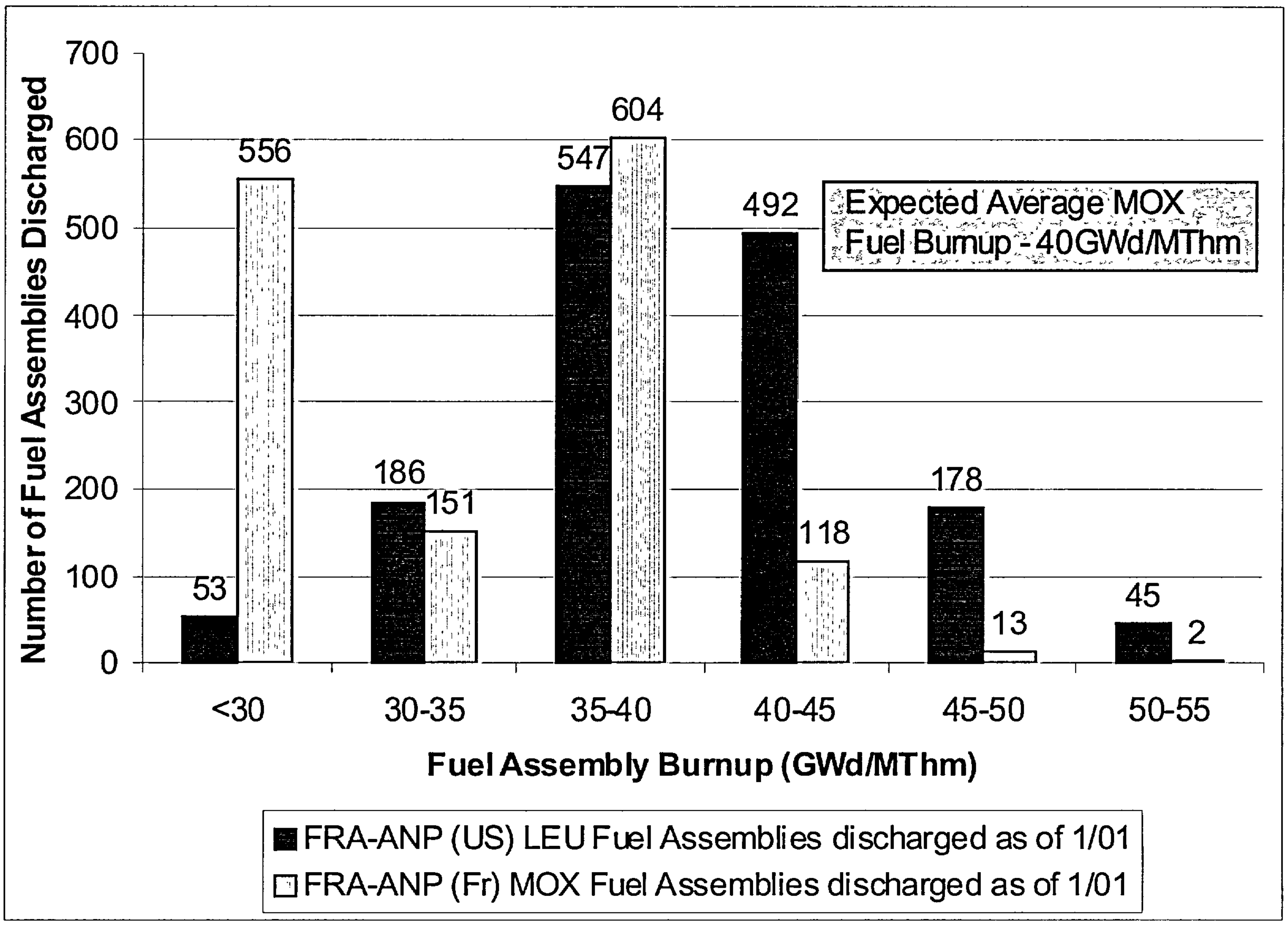
MOX Fuel Design Report – Experience Base



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MOX Fuel Design Report – Lead Assembly Program

> Lead assemblies will be fabricated, irradiated and examined as confirmation of the design and fabrication processes

- Addition of aqueous polishing ensures applicability of MIMAS process to WG material
- Adjustment of Master Mix ensures that MOX fuel from WG Pu is equivalent to RG MOX
- No data from lead assemblies required for model development or qualification of analytical models

MOX Fuel Design Report – Lead Assembly Program

> Planning basis is four (4) Mark-BW/MOX1 fuel assemblies
  - Limitations on availability of polished WG plutonium may limit the program to two (2) fuel assemblies – less desirable due to symmetry concerns, but consistent with previous confirmatory lead assembly programs

> All four mission reactors are the same design – irradiation in any one will meet the qualification requirements for batch implementation in all of the mission reactors

MOX Fuel Design Report – Lead Assembly Program

> Fabrication

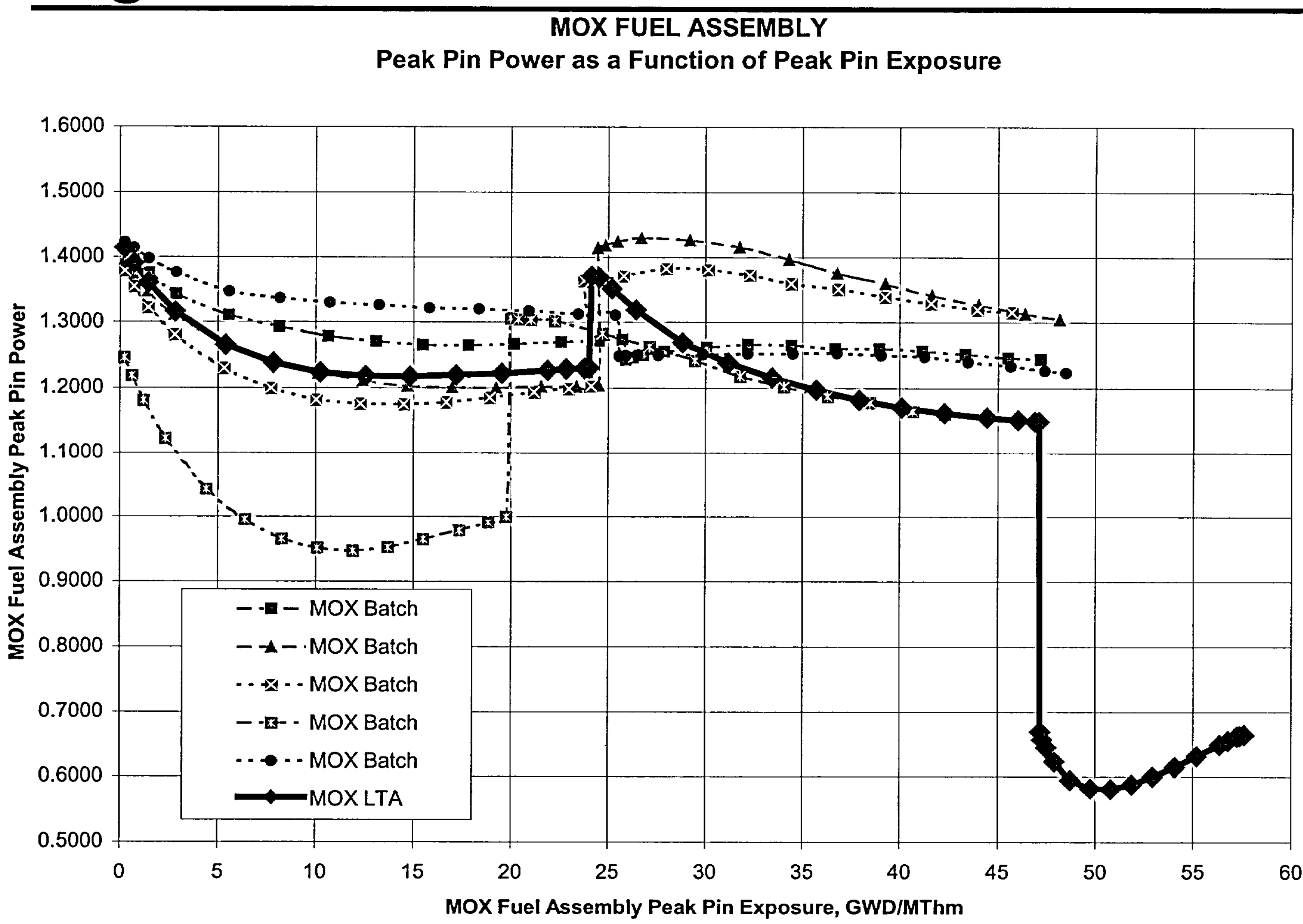
- Use WG Pu polished at LANL using prototypic processes and specifications consistent with the mission reactor fuel
- Use MIMAS process to produce pellets, consistent with mission reactor fuel
- Bundle fabrication using same design and processes as mission reactor fuel

MOX Fuel Design Report – Lead Assembly Program

> Irradiation Plan

- Irradiation in one of the four mission reactors for two (2) – 18 month cycles, consistent with mission reactor irradiation plans
- Core location will ensure relatively high power, yet remain non-limiting
- Two (2) cycle burnup projected to reach 47,000 MWd/MThm, consistent with the proposed pin burnup limit of 50,000 MWd/MThm
- Third irradiation cycle
  - Planned for one (1) assembly
  - To be performed to obtain data at higher burnup in support of future increases in burnup limits
  - Maximum three cycle pin burnup expected to exceed 57,000 MWd/MThm

MOX Fuel Design Report – Lead Assembly Program



MOX Fuel Design Report – Lead Assembly Program

> Fuel Examinations - Poolside

- Major components and fuel rods will be characterized
- Work scope
  - Visual
  - Fuel assembly growth
  - Crud
  - Fuel rod growth
  - RCCA drag
  - Fuel rod oxide
  - Fuel rod fission gas release
  - Water channel
  - Grid width
  - Grid oxide
  - Guide thimble oxide
  - Fuel assembly bow and distortion

MOX Fuel Design Report – Lead Assembly Program

> Fuel Examinations – Hot Cell

- Hot cell examinations
- Rod extractions after third cycle
- Work scope
  - Fission gas release
  - Fuel clad metallography
  - Fuel pellet ceramography
  - PCI
  - Burnup analysis
  - Burnup distribution

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MOX Fuel Design Report – Conclusion

> Extensive European experience is applicable to the Materials Disposition Program; weapons derived MOX is same as RG MOX due to:

- Use of aqueous polishing process
- Adjustment to Master Mix
- Application of consistent specifications

> Safe and reliable operation of fuel used for Material Disposition Program – Mark-BW/MOX1 – is confirmed

> The Mark-BW/MOX1 is acceptable for batch implementation up to a maximum fuel rod burnup of 50,000 MWd/MThm